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10 CFR 50.73

W3F1-2015-0072

August 21, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Subject: Licensee Event Report (LER) 2015-004-01, Emergency Feedwater System Flow Oscillations and LER 2015-005-01, Manual Reactor Trip due to Low Steam Generator Levels
Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

Dear Sir or Madam:

On June 3, 2015, Waterford Steam Electric Station, Unit 3 (Waterford 3) experienced a loss of main feedwater to both steam generators. Entergy is hereby submitting revisions to two LERs, 2015-004-01 and 2015-005-01, for events that occurred during this transient. The LERs were revised to add an abstract section per NUREG-1022.

LER 2015-004-01 provides details associated with a condition that resulted in a common cause inoperability of both trains of the Emergency Feedwater (EFW) System and could have impacted the past operability of both trains of the Emergency Feedwater System and the Atmospheric Dump Valves. It was determined that this condition is reportable pursuant to 10 CFR 50.73(a)(2)(i)(B), 10 CFR 50.73(a)(2)(v)(B), 10 CFR 50.73(a)(2)(vii) and 10 CFR 50.73(a)(2)(ix)(A). A follow up to LER 2015-004-01 is due by January 28, 2016 to provide the safety significance determination that is not yet complete.

LER-2015-005-01 provides details associated with a condition that resulted in the following System Actuations: Reactor Protection System Including Reactor Trip, Emergency Feedwater System, and Emergency Diesel Generators. It was determined that these conditions are reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A); specifically: 10 CFR 50.73 (a)(2)(iv)(B)(1), 10 CFR 50.73 (a)(2)(iv)(B)(6), and 10 CFR 50.73 (a)(2)(iv)(B)(8).

This report contains no new commitments. Please contact John P. Jarrell, Regulatory Assurance Manager, at (504) 739-6685 if you have questions regarding this information.

Sincerely,

JPJ/MMZ

Attachments: (1) LER 2015-004-01
(2) LER 2015-005-01

cc: Mr. Mark L. Dapas, Regional Administrator
U.S. NRC, Region IV
RidsRgn4MailCenter@nrc.gov

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Attachment 2

to

W3F1-2015-0072

Licensee Event Report 2015-005-01

NRC FORM 366 (02-2014)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 01/31/2017
LICENSEE EVENT REPORT (LER) (See Page 2 for required number of digits/characters for each block)		Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the FOIA, Privacy and Information Collections Branch (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to Infocollects.Resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.	

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4. TITLE
Manual Reactor Trip due to Low Steam Generator Levels

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
06	03	2015	2015 -	005	- 01	08	21	2015	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
100	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

LICENSEE CONTACT	TELEPHONE NUMBER *(Include Area Code)*
John Jarrell	5047396685

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
E	SN	LCV	W255	N	B	EA	RLY	A160	N

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On June 3, 2015, at 1705, the reactor was manually tripped due to low Steam Generator (SG) levels caused by a loss of the ‘A’ Main Feedwater (MFW) Pump. During the transfer of electrical buses, the ‘B’ train electrical buses failed to transfer to the Startup Transformer and were de-energized. The loss of ‘B’ power caused the loss of the remaining MFW pump and the Emergency Diesel Generator (EDG) on that side to automatically start and load the safety buses. The loss of MFW resulted in an Emergency Feedwater Actuation Signal (EFAS) with both SGs being fed by Emergency Feedwater (EFW) to restore level. The Emergency Operating Procedure (EOP) for a reactor trip due to the loss of MFW was entered. This event was caused by (1) the failure to identify the failure mechanism for repeated failures of non-safety related normal level control valves and (2) the relay’s timed contact sets not changing state due to an unknown equipment problem. Corrective actions include rebuilding valves with identified similar deficiencies, changing the frequency of the Preventive Maintenance (PM) procedures for level control valves, and establishing a relay replacement PM and identifying electrical control circuit system enhancements.

NARRATIVE

INITIAL CONDITIONS

The plant was in Mode 1 at 100% power with Reactor [RCT] Power Cutback (RXC) removed from service due to time in core life. Preventive maintenance was being performed on the Feed Heater Drain (FHD) [SN] 2C alternate level control valve [LCV]. Circulating Water Pump A was removed from service for maintenance on the traveling screen system.

EVENT DESCRIPTION

On June 3, 2015, at approximately 1700, the control room received multiple alarms due to FHD 2C experiencing high levels. Extraction steam (ES) [SE] to #2 heaters isolation isolated on high high level in the 2C FHD and the FHD 1C alternate drain valve [LCV] opened as designed. The control room entered the Off-Normal Procedure for Secondary System Transient. At 1702, the control room received annunciators for all three heater drain pump [P] low suction pressure and both MFW pumps [P] low suction pressure. The Control Room Supervisor (CRS) directed the Balance of Plant (BOP) operator to commence removing 100 MW's at a rate of 40 MW/min. At 1704, the heater drain pumps tripped on low suction pressure followed by the trip of MFW pump 'A'. The CRS entered the Off-Normal Procedure for a Rapid Plant Power Reduction. At 1705, the BOP operator reported that both SG [SG] levels were at 50% narrow range level and dropping rapidly (automatic reactor trip occurs at 27.4% narrow range). The CRS directed a manual reactor trip in anticipation of an automatic reactor trip from the Reactor Protection System (RPS) [JC] and entered the EOP for Standard Post Trip Actions. All rods fully inserted as designed and all required safety systems actuated as designed.

At the time of the manual reactor trip and main generator trip, the electrical buses [BU] [EA] [EB] were to transfer from the Unit Auxiliary Transformers (UATs) [XFMR] (supplied by the main generator) to the Startup Auxiliary Transformers (SUTs) [XFMR], which are fed from offsite power. The 'A' train electrical buses transferred as required, but the 'B' train buses failed to transfer to the Startup Transformer causing the loss of the 'B' train safety [EB] and non-safety buses [EA]. At 1705, EDG 'B' [DG] [EK] started and energized the safety buses as designed.

The loss of the 'B' non-safety buses caused the remaining FW pump to trip, resulting in a loss of all normal FW to both SGs. At 1707, the EFW [BA] system started on receipt of an EFAS [JE] and commenced feeding both SGs. The loss of the non-safety bus also caused a loss of 2 out of 3 circulating water pumps leading to a loss of main condenser vacuum. At 1706, the non-safety buses were manually transferred to the 'B' SUT but non-safety related bus loads remained de-energized.

EFW was feeding both SGs at 1715 when the BOP operator noted that the EFW backup flow control valves [FCV] exhibited wide, frequent oscillations. The BOP subsequently placed both trains of backup flow control valves in manual (this event is covered in LER-WF3-2015-004). At 1716, the crew entered into the EOP for Loss of Main Feedwater.

The non-safety train 'B' electrical buses were energized at 1817 from offsite power and the safety buses were transferred to offsite power at 1820. EDG 'B' was secured at 1824.

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At 2143, the Auxiliary Feedwater (AFW) [SJ] pump (non-safety related) was started and commenced feeding both SGs in preparation for securing the EFW Pumps. At 2231, EFW was secured with SG levels being maintained in their normal band using the AFW pump. On June 4, 2015, at 0224, the EFAS system was reset and the EFW system was placed in standby.

On June 4, 2015, at 0527, the plant exited the EOP for Loss of Main Feedwater and entered the Normal Operating Procedure for Plant Shutdown. The plant was stabilized in Mode 3 at normal operating temperature and pressure with heat removal being controlled on Atmospheric Dump Valves.

INVESTIGATION POST TRIP

The cause for FW heater 2C high level was that the 2C normal level control valve [LCV] stem and plug assembly had disengaged from the actuator stem. The stem threads were damaged and the lock pin was sheared. This caused the normal level control valve to fail closed. With the alternate level control valve removed from service for maintenance, all paths for removal of condensed water were unavailable, thus resulting in the high high level condition.

The failure of the ‘B’ UAT to transfer to ‘B’ SUT was determined to be a failure in the SUT ‘B’ time delay relay. Two sets of contacts would not change state when the relay was bench tested post removal. This failure mode would have prevented power to be transferred from UAT ‘B’ to SUT ‘B’ during a fast bus transfer. This relay is normally energized and the failure of the relay also explains why both the 6.9kV and 4kV feeder breakers from the SUT ‘B’ did not transfer.

SYSTEM DESCRIPTION

Feedwater Heater Drain System

The purpose of the extraction steam system is to supply steam to preheat the FW prior to entering the SGs. Preheating of the FW increases the cycle thermodynamic efficiency and minimizes thermal shock to the SGs. Extracting steam from the various stages of the turbine also removes moisture that collects as the enthalpy of the steam is reduced along the blade path.

The heater normal drains are arranged in a cascading fashion. The #1 heater drains to the #2 heater under normal conditions, with an alternate drainage valve that will automatically open to divert drainage to the condenser when heater water level is too high. The normal drains from the #2 heater are aligned to the suction of the heater drain pump which pumps the drain flow to the FW pump suction. Under abnormal high water level conditions, the drains from the #2 heater are automatically diverted to the condenser.

Emergency Feedwater System

The safety function of the EFW system is to provide a sufficient supply of cooling water to one or both SGs for the removal of decay heat from the reactor coolant system [AB] in response to any event causing low SG level coincident with the absence of a low pressure trip. The EFW system supplies this demand via three EFW pumps through two supply paths. Both supply paths are supplied with redundant instrument air operated flow control valves (FCVs) and isolation valves, all of which fail open on loss of air. The

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FCVs modulate EFW flow in response to SG level. These valves are designated as primary and backup. The FCVs change operating modes and setpoints based on changes in SG level indication.

Reactor Power Cutback System

The RXC system is actuated upon receiving coincident 2/2 logic signals indicating either large turbine load rejection or loss of one of two main FW pumps. The actuation logic initiates the insertion of the preselected pattern of CEAs. Subsequent insertion of other groups will automatically occur after the operator performs the immediate action of placing the control element drive mechanism control system control switch to the auto sequential position. On loss of a FW pump, a rapid turbine power reduction to 50 percent power is initiated, followed by a further reduction if necessary to balance turbine power with reactor power.

Electrical Distribution for Transfer of Unit Auxiliary Transformers to Startup Transformers

The automatic transfer uses a sequential design that requires the SUT supply breaker to receive its close signal from the opening UAT supply breaker. The nominal “dead time” for the sequential bus transfer is 3.3 cycles. The bus loads cannot distinguish between a "live bus" transfer and a "fast dead bus" transfer and will remain energized throughout the bus transfer process.

Bus transfer as a result of abnormal conditions is done automatically. Main generator lockout relays 86G1 and 86G2 will initiate a fast dead bus transfer of plant loads from the UATs to the SUTs. An automatic fast dead bus transfer will occur when the main generator lockout relays operate, as long as the trip was not caused by an overcurrent condition on the UAT feeder breakers, voltage is present at the SUTs, the UAT feeder breaker is open, and bus and incoming voltage/frequency are synchronized and in phase, as detected by synchronizing check relays.

REPORTABLE OCCURRENCE

Initial reportability (Message EN# 51116) was performed within 4 hours per 10CFR50.72(b)(2)(iv)(B), 4-hour Non-Emergency RPS Actuation (Scram), and 10CFR50.72(b)(3)(iv)(A), 8-hour Non-Emergency Specified Systems Actuation.

This condition is reportable under 10CFR50.73(a)(2)(iv)(A) (System Actuation); specifically: 10CFR50.73(a)(2)(iv)(B)(1), Reactor Protection System (RPS) Including Reactor Scram or Reactor Trip, due to a manual reactor trip prior to initiation of and automatic scram, 10CFR50.73(a)(2)(iv)(B)(6), PWR Auxiliary or Emergency Feedwater System, due to automatic initiation of EFW system and the addition of EFW to the SGs, and 10CFR50.73(a)(2)(iv)(B)(8), Emergency AC Electrical Power Systems, Including Emergency Diesel Generators, due to the automatic start of the EDG on loss of offsite power to the ‘B’ train electrical buses.

CAUSAL FACTORS

Failure of the Normal Level Control Valve

Root Cause: The root cause of the event is the failure to identify the failure mechanism for repeated failures of non-safety related normal level control valves.

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Contributing Causes: First contributing cause - Inadequate causal and technical evaluations for repeated failures of the non-safety related normal level control valves were accepted and tolerated by the site leadership team.

Extent of Condition:

This extent of condition is bounded by consideration of failures of normal and alternate level control valves for low, intermediate, and high pressure heaters. These valves are non-safety related valves and similar in purpose and design. The review identified that the normal feedwater intermediate pressure heater control valve (FHD-455A) is currently vulnerable to failure. The valve is not currently exhibiting lateral movement or chattering as was exhibited by the failed valve. Corrective action needed to address this condition is provided below. Several other normal level control valves for intermediate, low, or high pressure heaters with different size or similar design were identified that could fail closed while their alternate control valve is tagged out. The probability of occurrence of this happening is deemed medium with consequence from low to high. An interim action to address this is already in place to perform weekly schedule risk reviews for single point vulnerabilities and items that could result in an action requiring immediate shutdown therefore no further action is required at this time. Several other alternate level control valves of similar design were identified that could fail closed, or in a position other than closed, while their normal level control valve is tagged out. The probability of occurrence of this happening is deemed low with consequence from low to high. An interim action to address this is already in place to perform weekly schedule risk reviews for single point vulnerabilities and items that could result in an action requiring immediate shutdown therefore no further action is required a this time.

Corrective Actions:

(1) Change the frequency and due dates for the inspection and rebuild PMs for normal level control valves to be performed each refueling outage starting with the next refueling outage until the underlying failure mechanism for valves is determined and corrected.

(2) Develop and implement a "scheduling strategy" that contains expectations for scheduling online work to identify, assess and mitigate risks associated with trip/event initiators.

(3) Rebuild normal feedwater intermediate pressure heater control valve as soon as parts and plant conditions allow.

(4) Develop additional actions if needed after the failure mechanism of the failed level control valve is determined.

(5) Create a Trip/Event Initiator (TEI), requirement code in Asset Suite to identify work orders and PMs as TEIs to enable this information to be pulled into the P6 schedule layout.

<u>Failure of UAT 'B' to Auto Transfer to SUT 'B'</u>

Apparent Cause: The apparent cause for the Allen Bradley 700RTC11200U1 relay's timed contact sets C1-C2 and C3-C4 not changing state was due to an unknown equipment cause. Potential causes are failure of the relays C1-C2 and C3-C4 contact sets, or coil failure.

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Contributing Cause: A contributing cause for the lack of identification of the failure of the 4KVEREL2237 B RELAY (TL) prior to the loss of offsite power was due to a design change that was not adequate. A latent design deficiency (1997 plant change) did not take into account the observed failure mode where the two coil design allows one coil to fail thus preventing the dead bus transfer function (timed contacts) without actuating the contacts for the alarm circuit. This resulted in the alarm not being received prior to the failure to dead bus transfer from UAT 'B' to SUT 'B'.

Extent of Condition:

There are 38 Allen Bradley 700RTC relays installed at Waterford 3:

- Four relays are installed for fast bus transfer of the 4KV and 7KV busses. It was verified during plant shutdown that power was transferred from UAT 'A' to SUT 'A'.
- Thirty-three of these relays are installed in Engineered Safety Features Actuation System (ESFAS) applications. These relay failures are self-revealing (alarm or equipment start/stop/trip) and actions are in place to correct this condition.
- One relay was installed in core protection calculator panel D under temporary modification. The application is an alarm only circuit.

Corrective Actions:

(1) Generate an action request to establish a relay replacement PM task frequency for three years for four Allen Bradley 700RTC relays installed in the auxiliary relay cabinet.

(2) Fund an engineering study to make a recommendation for a relay to replace the existing fast bus transfer Allen-Bradley 700RTC series relays within auxiliary panel 4. The study should also pursue a new design that will provide proper alarm function should the coil for the control circuit fail.

(3) Funding an engineering study to make a recommendation for a relay to replace the existing fast bus transfer Allen-Bradley 700RTC series relays. The study should also pursue a new design that will provide proper alarm function should the coil for the control circuit fail.

SAFETY SIGNIFICANCE

Industrial Safety: There was no industrial safety significance associated with this issue.

Radiological Safety: There was no radiological safety significance associated with this issue.

Environmental Safety: There was no environmental safety significance associated with this issue.

Nuclear Safety: The actual consequence of this event was a manual reactor trip due to loss of MFW [SJ] pump 'A'. This unplanned reactor trip did not have any nuclear safety significance, based upon the operating shift's proper and timely diagnosis of the loss of MFW Pump 'A', and the initiation of a manual reactor trip. Since the operating shift was able to properly analyze the event and respond by manually tripping the unit, there are no potential consequences if an additional barrier failed or response actions

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were delayed. The RXC during this event was not in service. Because of this condition, the plant transient would have led to a SG low level condition resulting in an automatic reactor trip.

PREVIOUS OCCURRENCES

Waterford 3 Licensee Event Report history was reviewed.

LER WF3-2013-001-00: MFW regulating valve failed closed on loss of instrument air. This caused a manual reactor trip to be initiated.

There have been no previous occurrences where a reactor trip occurred due to a FW heater normal level control valve failing closed. Previous occurrences had only caused a reduction in plant power with unit remaining on line. This is based on a search of the Waterford 3 corrective action program database.

ADDITIONAL INFORMATION

Energy industry identification system (EIIS) codes and component function identifiers are identified in the text with brackets [ ].